Florida
Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

May 28, 1993
3F0593-12

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Subject: Response to NRC Information Notice 93-17

Reference: NRC to FPC letter, 3N0393-05, dated March 8, 1993

Dear Sir:

Information Notice (IN) 93-17, "Safety System Response to Loss of Coolant and Loss of Offsite Power," was issued to summarize NRC Staff concerns regarding the timing of loss of offsite power (LOOP) and loss of coolant accident (LOCA) and their effect on the emergency diesel generator loading logic. While IN 93-17 does not request licensee actions, Florida Power Corporation (FPC) believes that a response is necessary to avoid any mis-interpretation of the licensing basis and the design basis for Crystal River Unit 3 (CR-3).

The central theme of IN 93-17 apparently is the NRC's conclusion (page 2) that "[p]roperly designed safety systems will respond to all credible sequences of a LOOP and a LOCA." GDC-17 is cited as the basis for this position. It appears that the principal concern is a LOCA followed by a LOOP during load sequencing ("delayed LOOP"). The IN 93-17 concern is beyond the licensing basis and design basis for CR-3. CR-3's licensing/design basis scenario is a LOCA concurrent with a LOOP. The ECCS performance to comply with 10 CFR 50.46 and Appendix K is based upon this assumption. The plant-specific event described in IN 93-17 involved a potential occurrence of a LOOP event five minutes or more after a LOCA. In the cited event, the licensee even noted to the NRC that his original licensing/design bases did not include the delayed LOOP. Interestingly, IN 93-17 itself acknowledges that scenarios other than concurrent LOCA/LOOP may not be included in the licensing basis for most plants.

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FPC will take no further action on this notice based on the following reasoning.

1. The NRC has previously considered the delayed LOOP scenario. In December 1983, NUREG-0933, "A Prioritization of Generic Safety Issues," designated the scenario as Generic Issue 17, "Loss of Offsite Power Subsequent to a LOCA." The latest issue of NUREG-0933, dated August 1992, does not list the issue. The NRC apparently concluded the probability of occurrence of delayed LOOP is not significant enough warrant regulatory action.

2. In NUREG/CR-4893, "Technical Report for Generic Issue 135: Steam Generator and Steam Line Overfill Issues," May 1991, the NRC stated:

   3.11.4. Delayed LOOP

   Consideration was given to the probability of occurrence of a delayed LOOP as developed in GI 17. The cumulative probability of delayed LOOP and failure of an operator to pick up the loss of power is 2×10^{-6} at one hour after the LOCA. The probability of occurrence of a delayed LOOP is low enough not to be considered. Therefore, this is no longer considered in the design basis LOCA. It was similarly dismissed by the NRC Commissioners during the rulemaking for 10 CFR Appendix K in 1972 and therefore should not be considered in the design basis SGTR event.

3. CR-3 was not designed in accordance with GDC-17. Since CR-3 was issued a construction permit prior to May 21, 1971, CR-3 is designed to those general design criteria specified in FSAR Chapter 1. The Commission recently stated that the current 10 CFR 50, Appendix A, General Design Criteria do not apply to plants like CR-3, on the basis that pre-GDC plants were "evaluated on a plant specific basis, determined to be safe, and licensed by the Commission." This position was stated in Staff Requirements Memorandum (SECY-92-223), "Resolution of Deviations Identified During the Systematic Evaluation Program," September 18, 1992.

The notice should be withdrawn until appropriate review by the Committee to Review Generic Requirements (CRGR) in accordance with NRC guidelines. As written, the notice could be interpreted by NRC inspectors as suggesting that present plant licensing/design bases are not in accordance with NRC design criteria. That is simply not the case for CR-3. CRGR should request the NRC staff to provide more information regarding the history of the issue to prevent utilities from expending unnecessary resources on an issue that has been resolved previously.

Sincerely,

P. Beard Jr.

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

PMB/JWT

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager